NON-PUBLIC?: N
ACCESSION #: 9308190119
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Salem Generating Station - Unit 1 PAGE: 1 OF 06

DOCKET NUMBER: 05000272

TITLE: Rx Trip On 14 Steam Generator Low Level Coincident With SF/FF Mismatch & TS Noncompliance.
EVENT DATE: 07/19/93 LER #: 93-013-00 REPORT DATE: 08/12/93

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 097

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(i) and 50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: M. J. Pollack - LER Coordinator TELEPHONE: (609) 339-5163

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: JG COMPONENT: RLY MANUFACTURER: W120
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On 7/11/93, at 2038 hours, the Unit experienced a reactor/turbine trip signal due to steam flow/feed flow mismatch coincident with low level in 14 Steam Generator (SG). This trip resulted from closure of 14 SG Feedwater Regulating Valve, 14BF19. Shutdown had been in progress per Tech. Spec. 3.3.2.1 Action 13, due to inoperability of part of the Solid State Protection System (SSPS). It was determined that relay BD601 was inoperable requiring its removal. During relay removal, the 125 VDC lead to the component normal actuation circuit was removed causing associated components (solenoid valves for the 14BF19 valve and its bypass valve, 14BF40) to deenergize. The 14BF19 valve closed per design (14BF40 was already closed) causing the trip. The root cause is personnel error.
Due to inattention to detail, the maintenance supervisor did not fully assess the affect of disconnecting the BD601 wiring. Corrective discipline has been taken with the supervisor. This event will be reviewed with applicable personnel. Relay BD601 was replaced. On

8/5/93, management review of the reactor trip event investigation determined that the SSPS slave relay operability determination was not diagnosed accurately on 7/11/93, at 0530 hours, and that expeditious troubleshooting was not initiated as appropriate. The cause of this event is personnel error. Appropriate corrective discipline will be taken with the operations personnel involved. This event will be reviewed with all licensed operator personnel.

END OF ABSTRACT

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as {xx}

IDENTIFICATION OF OCCURRENCE:

Reactor Trip On 14 Steam Generator Low Level Coincident With Steam Flow/Feed Flow Mismatch and Technical Specification Noncompliance

Event Date: 7/11/93
Discovery Date (Technical specification Non Compliance): 8/05/93
Report Date: 8/12/93

This report was initiated by Incident Report No. 93-302. it is required per Code of Federal Regulations 10CFR50.73(a)(2)(iv) and 10CFR50.73(a)(2)(i)(B).

CONDITIONS PRIOR TO OCCURRENCE:

Mode 1 Reactor Power 97% - Unit Load 1100 MWe

On July 11, 1993, shutdown to HOT STANDBY (MODE 3) was in progress per Technical Specification (T/S) 3.3.2.1 Action 13, due to inoperability of Solid State Protection System (SSPS) Feedwater Isolation Circuit Train "B" {JG}. The Action Statement had been entered at 1730 hours. It requires the Unit to be placed in Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours.

DESCRIPTION OF OCCURRENCE:

On July 11, 1993, at 2038 hours, the Unit experienced a Reactor/Turbine Trip signal due to steam flow/feed flow mismatch coincident with low level (25%) in 14 Steam Generator (SG) . The mismatch and 14 SG low level resulted from unplanned closure of 14 SG Feedwater Regulating Valve, 14BF19 {SJ}.

Emergency Operating Procedure EOP-TRIP-1 was entered, the Auxiliary Feedwater pumps automatically started on low SG levels, and Main Steam was manually isolated to minimize Reactor Coolant System (RCS) {AB} cooldown. The Unit was stabilized in MODE 3. The Nuclear Regulatory Commission (NRC) was notified of shutdown initiation and automatic actuation of the Reactor Protection System (RPS) {JC} per 10CFR50.72 (b)(1)(i)(A) and 10CFR50.72(b)(2)(ii).

On August 5, 1993, as a result of management review of the reactor trip event investigation, it was determined that the operability determination of the SSPS slave relay was not diagnosed accurately on July 11, 1993 and that expeditious troubleshooting was not initiated

DESCRIPTION OF OCCURRENCE: cont'd

as is appropriate. This does not comply with the guidance provided by NRC Generic Letter 91-18, dated November 7, 1991.

At the time of the trip, a shutdown was in progress per T/S 3.3.2.1 Action 13 due to inoperability of SSPS Feedwater Isolation Circuit Train "B". The SSPS Train B circuit problem was initially discovered at 0530 hours (see Analysis of Occurrence) on July 11, 1993. The Action Statement had not been entered due to Operations personnel belief that the test circuit had failed, not the SSPS circuit output relay. It was not until maintenance identified the output relay failure, approximately twelve (12) hours later, that the Action Statement was entered.

NRC Generic Letter 91-18, addresses operability considerations. Section 4.0, "Background" states:

"... The determination of operability for systems is to be made promptly, with a timeliness that is commensurate with the potential safety significance of the issue. If the licensee chooses initially not to declare a system inoperable, the licensee must have a reasonable expectation that the system is operable and that the prompt determination process will support that expectation. Otherwise, the licensee should immediately declare the system or structure inoperable. Where there is reason to suspect that the determination process is not, or was not prompt the Region may discuss with the licensee, with NRR consultation as appropriate, the reasoning for the perceived delay. ..."

ANALYSIS OF OCCURRENCE:

On July 11, 1993, at approximately 0530 hours, slave relay testing (procedure S1.OP-ST.SSP-0010(Q), "ESF-SSPS Slave Relay - Train B") was in progress. The surveillance was stopped when a problem occurred in obtaining a test meter reading during "Slave Relay K601 - Safety Injection" circuit testing. Based upon initial print review the problem appeared to be in the test circuit portion of the output relay, not the output relay portion of the circuit. The test circuit is independent of normal SSPS function. Test circuit problems, independent of the SSPS circuit have been encountered several times recently. Therefore, the SSPS was not declared inoperable. A work request was initiated to investigate the concern and it was decided to troubleshoot the problem when an SSPS qualified supervisor was scheduled to arrive later during day shift. The supervisor initiated investigation that afternoon of the SSPS surveillance test circuit concern. At 1730 hours the maintenance supervisor informed operations that the test circuit had not failed and that the surveillance results showed an SSPS circuit failure, based on review of the circuit prints. Specifically, buffer relay BD601 for feedwater isolation was apparently inoperable.

ANALYSIS OF OCCURRENCE: (cont'd)

To support repair, it was necessary to remove the BD601 relay. During removal, the technician removed leads from the BD601 relay. One of the BD601 terminals had two leads (on a common screw). These leads supply 125 VDC to the component normal actuation circuit and to the test (bypass) circuit connection. The components associated with this actuation circuit are the solenoid valves for the 14BF19 valve and its bypass valve, 14BF40. Separating the two (2) 125VDC leads, resulted in loss of 125 VDC to the solenoid valves. The 14BF19 valve then closed per design (14BF40 was already closed) causing the 14 SG steam flow/feed flow mismatch coincident with low S/G level reactor trip.

Continued investigation of the Slave Relay K601 - Safety Injection test failure confirmed that the BD601 relay had failed due to an open operate coil. This failure would have prevented closure of the 14BF19 and 14BF40 valves during a Feedwater Isolation signal following a Train B Safety Injection signal. investigation is continuing to determine the specific cause of the failed BD601 relay.

The RPS reactor/turbine trip signal, on steam flow/feed flow mismatch coincident with low level, is anticipatory. Its function is to prevent a loss of heat sink capability by sensing conditions which could eventually result in a dry steam generator. By tripping the reactor prior to reaching the low-low level trip setpoint, the required starting time and capacity requirements for the Auxiliary Feedwater System (AFW) {BA} are reduced; thereby, minimizing the thermal transient on the SGs and the RCS.

The RPS functioned as designed and the heat sink was maintained during this event. Following the reactor trip, the AFW flow indicator to 12 SG did not respond properly. Per EOP-TRIP-1, operators started 13 AFW pump. No adverse impact occurred due to the failed indicator or start of the 13 AFW pump. Also, Main Steam Isolation was initiated in accordance with EOP-TRIP-2 due to excessive RCS cooldown. Reduction in T sub avg, requiring main steamline isolation, has been experienced during other reactor trips (e.g., Unit 1 LER 272/93-004-00). The start of the 13 AFW Pump contributed to the cooldown experienced in this event.

APPARENT CAUSE OF OCCURRENCE:

Reactor Trip Event

The root cause of this event is personnel error.

To support troubleshooting to determine whether the test circuit had failed or an actual SSPS function was impaired, a maintenance supervisor (qualified on SSPS) reviewed the circuit diagrams in relation to the observed readings. He concluded that the problem was most likely in the portion of the circuit associated with the BD601 relay. After notifying the shift of the SSPS circuit failure, he

APPARENT CAUSE OF OCCURRENCE: (cont'd)

continued with a plan to support BD601 circuit troubleshooting in accordance with procedure SC.IC-GP.ZZ-0006(Q), "Controls Equipment - Troubleshooting". Due to inattention to detail, he did not fully assess the affect of disconnecting the BD601 wiring.

Technical Specification Event

The root cause of not fully complying with Technical Specification 3.3.2.1 is personnel error. The decision that the failure observed during the surveillance was in the test circuit was not correct. With qualified SSPS maintenance personnel not immediately available, an inappropriate decision to delay troubleshooting was made.

PRIOR SIMILAR OCCURRENCES:

Reactor Trip Event

RPS signal actuation on steam flow/feed flow mismatch coincident with low SG level has occurred in the past. Two (2) such events, dated February 18, 1993 (reference LER 272/93-005-00) and February 6, 1989 (reference LER 272/89-007-00), involved personnel error. However, the specific circumstances surrounding the cause of those events differ substantially with this one.

A similar set of causal factors did lead to a reportable event on February 9, 1991 (reference LER 272/91-003-01). That event involved an unplanned Technical Specification 3.0.3 entry due to two (2) steam flow channels on one main steam line being made inoperable. It was also due to inadequate planning by maintenance supervision. The past event was viewed as an isolated occurrence. corrective action was limited to discipline and department personnel review.

Technical Specification Event

A review of prior Technical Specification non compliance events was

conducted. A similar event for ones involving lack of prompt investigation was not identified.

SAFETY SIGNIFICANCE:

These events did not affect the health or safety of the public.

Although the Train B Safety Injection Signal would not have caused a 14 SG feedwater isolation signal, the Train A signal was available. Therefore, in the event of an actual plant transient requiring safety injection and feedwater isolation, it would have occurred.

Also, other than the 12 SG AFW flow indication which did not adversely affect plant response, equipment required to function following the reactor/turbine trip functioned per design. Diverse indications were available and operable to determine the AFW flow to 12 SG.

CORRECTIVE ACTION:

This event has been reviewed by Maintenance Department management. Corrective disciplinary action has been taken with the supervisor involved.

The circumstances surrounding this event will be reviewed with applicable Maintenance Department personnel.

Maintenance procedure SC.IC-GP.ZZ-0006(Q), "Controls Equipment - Troubleshooting" has been revised (as of August 3, 1993). It now details the level of troubleshooting plan review required based on risk assessment (i.e., safety or plant transient). Had this procedure been implemented prior to this event, the BD601 troubleshooting plan would have required system engineering and maintenance management involvement as a minimum.

The BD601 relay was replaced. Investigation to determine the specific cause of the BD601 relay failure is continuing.

The 12 SG AFW flow indication failure was repaired. The transmitter was found to be out of calibration generating false signals.

To address the post trip T sub avg reduction concerns corrective actions are being implemented, as discussed in prior LERs (e.g., 272/93-004-00).

Operations Department management has reviewed the circumstances surrounding the lack of expeditious troubleshooting event. Corrective disciplinary action will be taken with the operations personnel involved.

The circumstances surrounding the Technical Specification non compliance event and Generic Letter 91-18 will be reviewed with Operations Department licensed personnel. A night order book entry has been made regarding the appropriate action statement entry, if the readings in the slave relay testing procedures are abnormal in any switch position.

The circumstances surrounding the Technical Specification non compliance will be reviewed by the Nuclear Training center for licensed operator training program changes as applicable.

The Operations procedures for slave relay functional testing (both units) will be reviewed and revised as appropriate.

General Manager -
Salem Operations
MJP:pc
SORC Mtg. 93-075

ATTACHMENT 1 TO 9308190119 PAGE 1 OF 1

PSE&G

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038

Salem Generating Station

August 12, 1993

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Dear Sir:

SALEM GENERATING STATION
LICENSE NO. DPR-70
DOCKET NO. 50-272
UNIT NO. 1

LICENSEE EVENT REPORT 93-013-00

This Licensee Event Report is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR 50.73(a)(2)(iv) and 50.73(a)(2)(i)(B). This report is required to be issued within/ thirty (30) days of event discovery.

Sincerely yours,

C. A Vondra
General Manager -

Salem Operations

MJP:pc

Distribution

The power is in your hands.

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